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THERMAL-HYDRAULIC RESPONSE OF A REACTOR CORE FOLLOWING LARGE BREAK LOSS-OF-COOLANT ACCIDENT UNDER FLOW BLOCKAGE CONDITION

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ABSTRACT

Since the revision of the requirements to consider the effect of fuel burnup on emergency core cooling system performance was proposed, flow blockage in reactor core has been one of the important issues in the thermal-hydraulic analysis of loss-of-coolant accident (LOCA). The present paper describes how much flow blockage would be expected following a large break LOCA based on the actual nuclear design data including the power and burnup of the fuel rods. A system thermal-hydraulic code, MARS-KS, is used for calculation where the burnup specific data of the fuel rods is supported by a fuel performance code, FRACON3. To recover the weakness of the system code in which the flow blockage under multiple rods configuration cannot be automatically simulated in hydraulic calculation, a special modelling scheme is developed and applied to the calculation. The effect of flow blockage on the thermal-hydraulic response of the reactor core is also discussed. To compensate for the uncertainty of the present flow blockage model, additional calculations are attempted for a wide range of the level of blockage.

Keywords: Effect of Fuel Burnup, Flow Blockage in Reactor Core, Hydraulic Modelling of Swelling and Rupture of Cladding, Large Break LOCA, MARS-KS Code.

1 INTRODUCTION

Flow blockage of the reactor core can occur due to swelling and rupture of the cladding of fuel rods following a large break loss-of-coolant accident (LBLOCA) in nuclear power plants (NPPs) and has been required to consider for the analysis of Emergency Core Cooling System (ECCS) performance [1]. Swelling of the cladding is reasoned for the pressure difference between gap and the fluid outside the cladding induced by a significant depressurization following a LOCA, and the degradation of heat transfer to the fluid outside the cladding. Swelling can lead to an excessive plastic deformation and eventually to rupture of cladding when exceeding a certain level of deformation [2]. Accordingly, the level of blockage is dependent on the swelling and rupture over the core.

As the requirement of the consideration of the effect of fuel burnup on ECCS performance was recently proposed [3], the flow blockage has been one of the important issues in the thermal-hydraulic analysis, because it can be severe for the high burnup fuel, and it would cause an unexpected change in thermal-hydraulic response of the core following a LOCA.

The present paper describes firstly how much flow blockage would be expected following a LBLOCA based on the actual nuclear design data including the distributions of power and burnup. For this purpose, a system thermal-hydraulic code, MARS-KS [4], was used and the burnup dependent data of the fuel rods was supported by a fuel performance code, FRACON3 [5]. Since the flow blockage occurred during the transient is not directly simulated in hydraulic calculations of the current system codes, such as MARS-KS and RELAP5 [6], when adopting multiple fuel rods, a special modelling scheme to address this problem was developed in the present study. Use of multiple fuel rods modelling was found in some researches [7], however, the specific treatment of the flow blockage using the system thermal-hydraulic code was not clearly discussed.

Secondly the effect of flow blockage on the thermal-hydraulic response of the reactor core was discussed. Generally, it has been known that the blockage is to improve heat transfer both

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upstream and downstream of the blocked region and that a blockage up to 60% has an insignificant effect on core heat transfer [2]. However, it is still questionable whether those results are valid for an actual LBLOCA due to several differences including flow bypass in the core between the experiments [2] and the actual NPP. Accordingly, the uncertainties of the code and calculation of the swelling and rupture in the actual reactor core should be considered to cover sufficiently the effect of those differences. For this aspect, additional calculations were attempted for a wide range of level of flow blockage to cover such uncertainties.

2 MODELLING SCHEME

2.1 Hydraulic modelling

Figure 1 shows one of the four quadrants that make up the reactor core, which is composed of several fuel assemblies (FAs) having fuel rods. Each FA and each fuel rod have a different power and different burnup from the neighbouring FAs and fuel rods.

In the existing safety analysis of the NPP, LBLOCA has been calculated using a lumped system model, in which the core was modelled by one average channel and one hot channel. The former simulated all the FAs except one hot FA and the latter simulating one hot FA with one hottest rod. Moreover, all the inputs were prepared for the condition of the begin-of-life (BOL) of the first cycle, in which burnup of all the fuel rods were less than 5 Giga-Watts Days per Metric Tons of Uranium (GWD/MTU). Accordingly, the results of the calculation did not properly address the actual fuel rod behaviour at higher burnup conditions. As the burnup increases, the thermal conductivity of the pellet degrades, accordingly, the fuel stored energy increased at the same fuel power, and the peak cladding temperature (PCT) following a LOCA may increase. Therefore, the calculation considering the higher burnup has been strongly requested since other performance parameters might also be affected by the burnup.

To establish a regulatory position on these issues, a modelling scheme of multiple channels with multiple fuel rods reflecting the actual core design was developed by the author in their previous study [8]. As shown in the lumped system model of Fig. 1, several hydraulic channels can be used. The number of hydraulic average channel is determined by grouping of the FAs. An FA power ratio or an FA burnup level can be a grouping criterion. Each group has

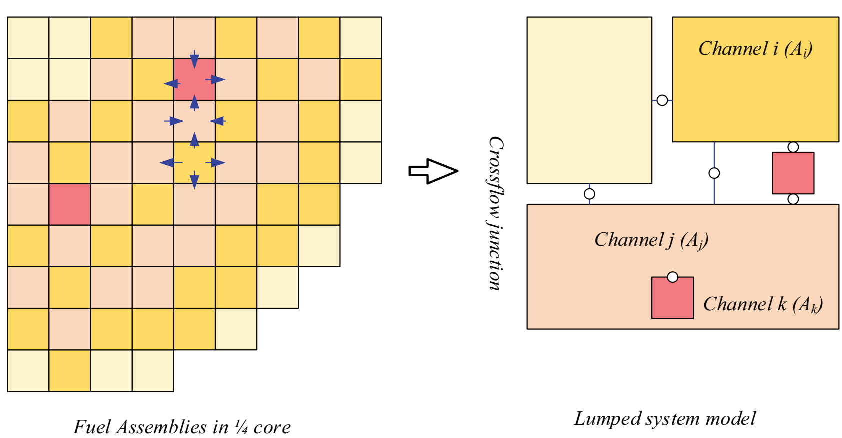


Figure 1: Reactor core modelling scheme.

one average channel with/without a hot channel and it is inter-connected with crossflow junctions. Several heat structures simulating the fuel rods within the group and the FA and special rods to be investigated can be allocated.

2.2 Flow blockage modelling

Figure 2 shows a concept of the flow blockage model. Consider the fuel rods in the FA k was geometrically changed by swelling and rupture following an accident. Then the flow area change of the FA k can be as follows:

$$F_k = A_k - A_{0k} = \sum_{i=1}^{n_k}\left(a_{k,i} - a_{0k,i}\right) = \sum_{i=1}^{n_k}\pi(r_{0k,i}{}^2 - r_{k,i}{}^2), \tag{1}$$

where n_k denotes number of fuel rod per FA and subscript 0 mea initial state. If the rod was ruptured, then $a_{k,i} = 0$ is applied. The radius of the fuel rod cladding is calculated by the cladding deformation and rupture model [9] of the MARS-KS code.

Since eqn (1) is a sum of changes of area of all the rods in an FA and this equation will be applied to all the FA, tens of thousands of calculations on cladding deformation are needed. For convenience, it is assumed that the range of radial power peaking factor (Fxy) and burnup of the fuel rods can be subdivided into M and N sections, respectively, and the number of fuel rods corresponding to each section can be counted. Generally, the hydraulic channel i has several fuel rods, thus, the change of flow area of the channel i is as follows:

$$G_i = A_i - A_{0i} = \sum_{m=1,n=1}^{m=M,n=N} D_{i,m,n} f_{m,n}. \tag{2}$$

where, m, n denote the section number of power and burnup and their maximums are M and N, respectively. $D_{i,\,m,\,n}$ means a matrix of the number of fuel rod at the m-th burnup interval and the n-th peaking factor interval, and belonging to i channel. And the area change of the fuel rod, $f_{m,\,n}$, can be calculated as follows:

$$f_{m,n} = a_{m,n} - a_{0m,n} = \pi\left(r_{0,m,n}{}^2 - r_{m,n}{}^2\right), \tag{3}$$

Therefore, one can calculate the level of flow blockage due to swelling and rupture by (1) the information of initial cladding radius, power, and burnup for the entire fuel rods, (2)

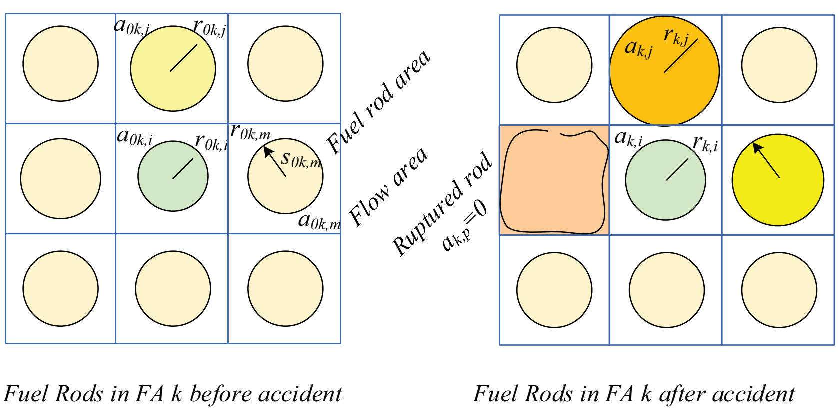


Figure 2: Concept of fuel rod swell and rupture model.

number of fuel rods at each section of power and burnup, and (3) cladding outer radius following an accident calculated by MARS-KS code. The level of the flow blockage can be implemented into the area change of the hydraulic channel using a SERVO valve model [4] of the MARS-KS code. The normalized value of G_i of eqn (3) is used for that model.

3 PLANT CORE MODELING

The flow blockage modelling described above was applied to the core at the end-of-life (EOL) of Cycle 2 of Shinkori Unit 3 [10], the first plant of the advanced power reactor (APR) 1400. The multiple fuel rod modelling scheme was also applied. Based on the information provided by the fuel vendor [11], the reactor core was modelled by two groups (one average channel and one hot channel per each group) as shown in Fig. 3.

In total, 30 heat structures were used to represent the fuel rods as shown in Table 1 and Fig. 4. Some of the rods were for the flow blockage calculation (representative rods). Ranges

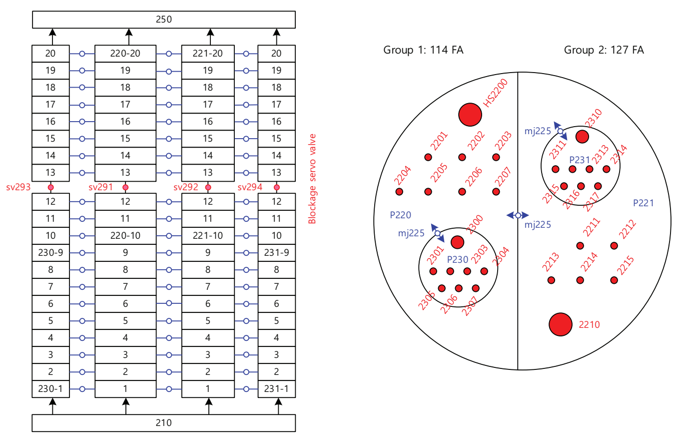


Figure 3: MARS-KS model for Reactor Core and Heat Structures.

Table 1: Radial peaking factors and burnup of heat structures.

HS number	Description	Radial peaking factor, *Fxy*	Burnup (GWD/ MTU)
2200	Average rods in Group 1	1.1239	21.4
2201~2207	Representative rods to Fxy/ Burnup sections	1.1, 1.2, 1.3, 1.2 1.3, 1.4, 1.45	25, 25, 25, 35, 35, 35, 35
2210	Average rods in G-2	0.8846	34.06
2211~2215	Representative rods	1.1, 1.2, 1.1, 1,2, 1,1	35, 35, 45, 45, 55
2300	Average rods in hot FA, G-1	1.3577	22.97
2301~2307	Representative rods and hot rods	1.1, 1.4, 1.45, 1.54, 1.54, 1.54, 1.4	15, 25, 25, 10, 20, 30, 40
2310	Average rods in hot FA, G-2	1.0381	30.5
2311~2317	Representative rods and hot rods	1.1, 1.1, 1.2, 1.48, 1.28, 1.25, 1.18	25, 35, 35, 35, 40, 45, 50

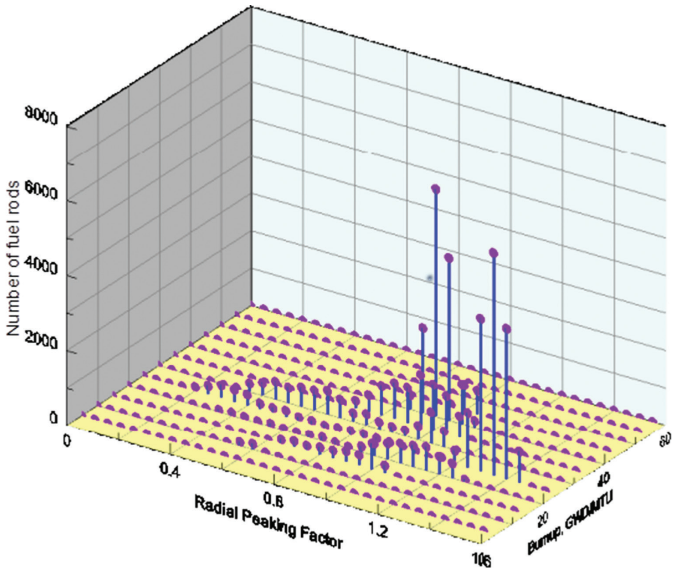


Figure 4: Burnup, radial peaking factors and number of rods for each heat structure.

of burnup and *Fxy* were 10–55 GWD/MTU and 0.1–1.45 and discretized with intervals of 10 GWD/MTU and 0.1, respectively.

4 RESULT AND DISCUSSION

4.1 Base case

Figure 5 shows the calculated cladding radii of all the simulated fuel rods. As shown in the figure, the rods having *Fxy* less than 1.1 even did not swell while others swelled and/or ruptured. From the results, one can find the high powered rods in the Group 1 were ruptured while the hot rods in Group 2 remained non-ruptured. This was due to the difference in fluid temperature outside the fuel rods, which was caused by the difference of *Fxy* of hot FA.

The calculated flow blockage at all the hydraulic channels using those results is shown in Fig. 6. As shown in the figure, flow area reductions were 10% and 86% at the average channel and the hot channel of Group 1, respectively, while both were less than 1% in Group 2. Such a high blockage at the hot channel of Group 1 was due to the conservatism in the assumption of complete blockage of the flow area belonging to the ruptured rod, the division of the *Fxy* and burnup sections, the estimation of the number of rods for each section, and the use of conservative core decay model, ANS73 model [12], built-in the MARS-KS code.

Figure 7 shows a comparison of the cladding temperatures at 17 fuel rods of interest. The presented calculation shows a significant difference in cladding heatup behaviour between the fuel rods in the average channel and the rods in the hot channel, and also difference between Group 1 and Group 2. Also the decrease of cladding temperature by return to nucleate boiling during blowdown period was found in Group 2, which is due to the relatively lower power. At the time of significant increase of blockage during 50–75 seconds (in Fig. 6),

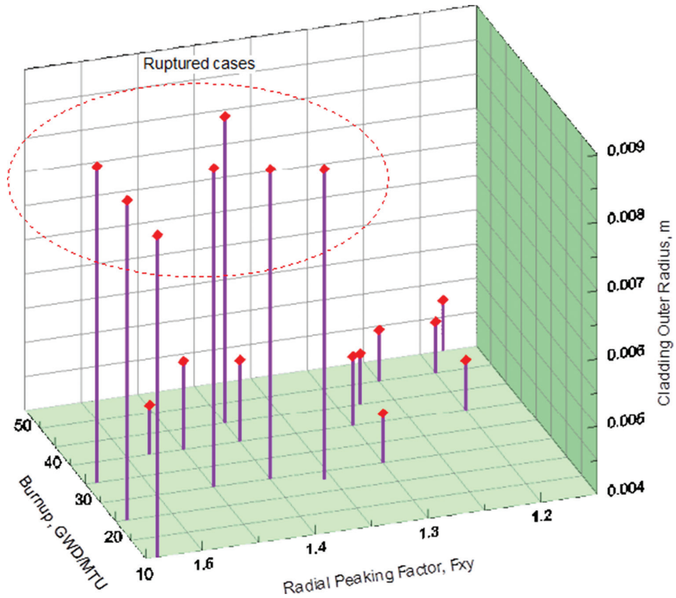


Figure 5: Calculated cladding radius.

one can find the decrease and re-increase in cladding temperature for the fuel rods at the hot channel. This can be regarded as an effect of flow blockage, i.e. a cooling by flow acceleration and a restriction of reflooding. One can find the effect of high blockage at the hot channel of Group 1, i.e. the heating up behaviour and delay in quenching time.

Figure 8 shows comparison of the calculated PCTs during blowdown phase and reflood phase versus *Fxy*, respectively. It is shown that PCTs increased as the *Fxy* increased and the burnup has a tendency to expand this increase. This trend is valid in PCT during the reflood phase with the exception of fuel rods in the hot FA of Group 2. The fuel rods have experienced the return to nucleate boiling, as shown in Fig. 7, which led to a quietly low reflood PCT.

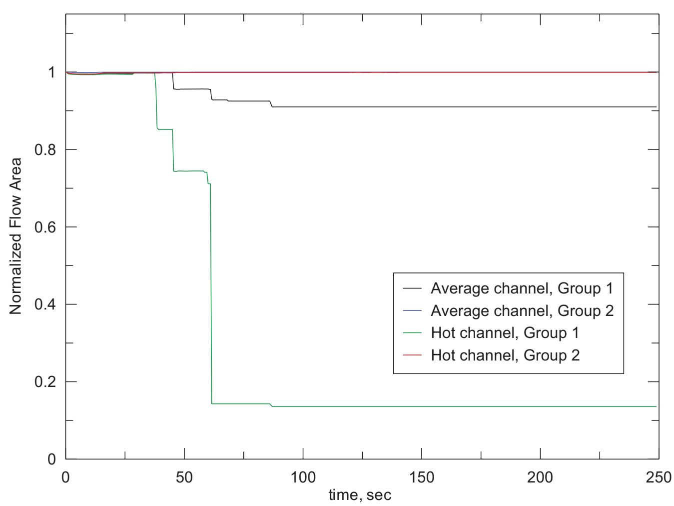


Figure 6: Calculated flow area.

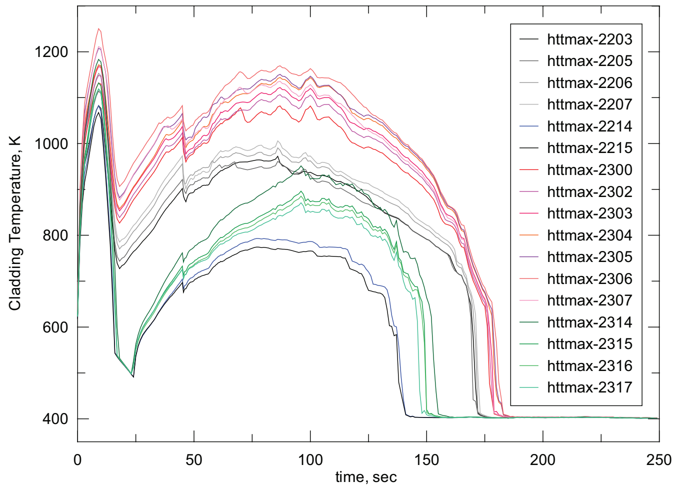


Figure 7: Calculated cladding temperatures.

Figure 9 shows the calculated oxide thickness of all the fuel rods. Similar to PCT, the transient oxide thickness increases as the *Fxy* and the burnup increase, and the maximum was less than 6 μm.

4.2 Sensitivity case

In the base case, the calculated change of cladding outer radius was directly implemented into the flow area change. As discussed, the estimate of blockage may involve uncertainties in several parameters such as the swell and rupture model of the code, the discretization of power and burnup, and the determination of the number of the fuel rods at each section. To compensate for those uncertainties, a sensitivity study was attempted regarding the several levels of blockage from 0% to 86 % of the flow area of hot channel.

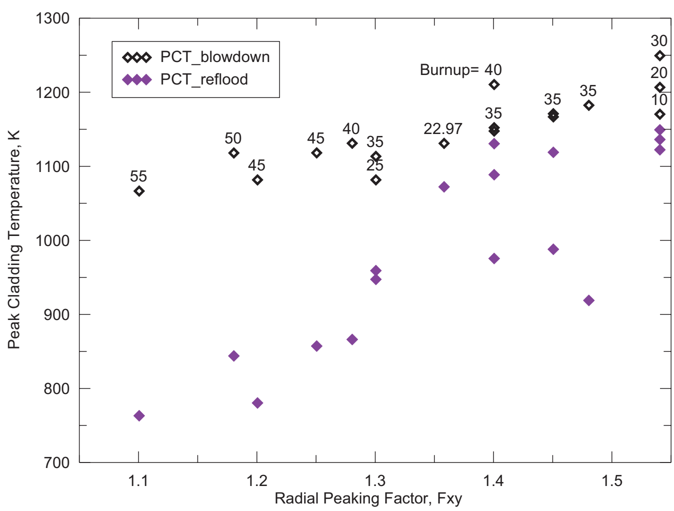


Figure 8: PCT versus *Fxy*.

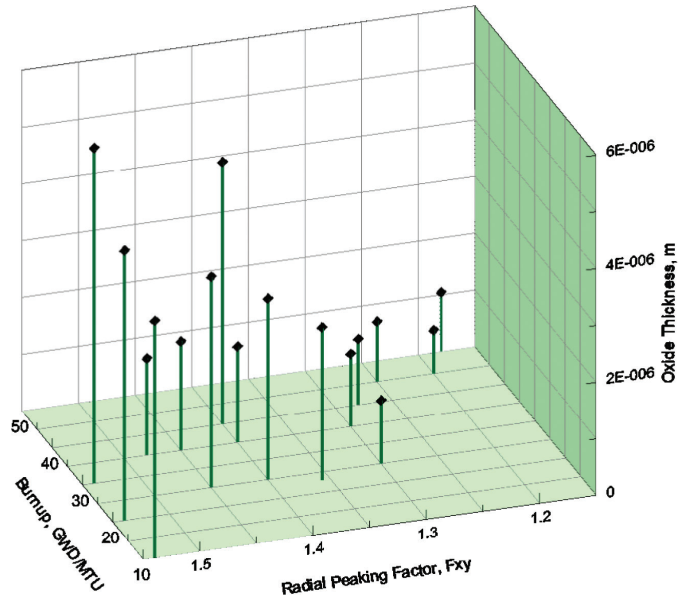
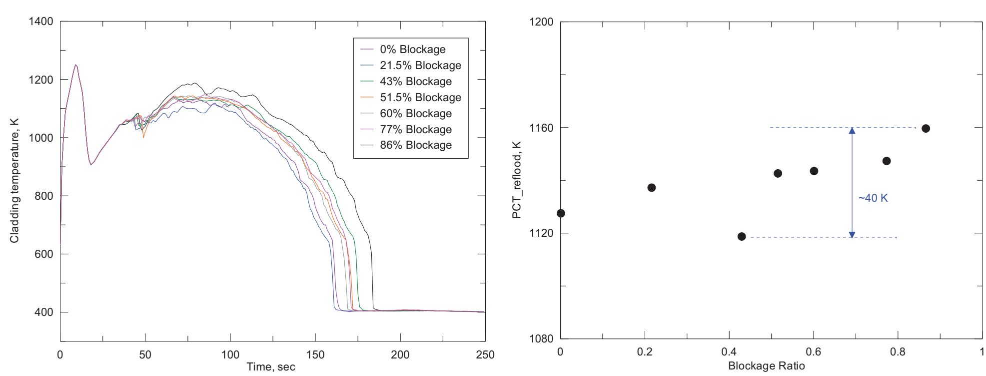


Figure 9: Calculated oxide thickness.

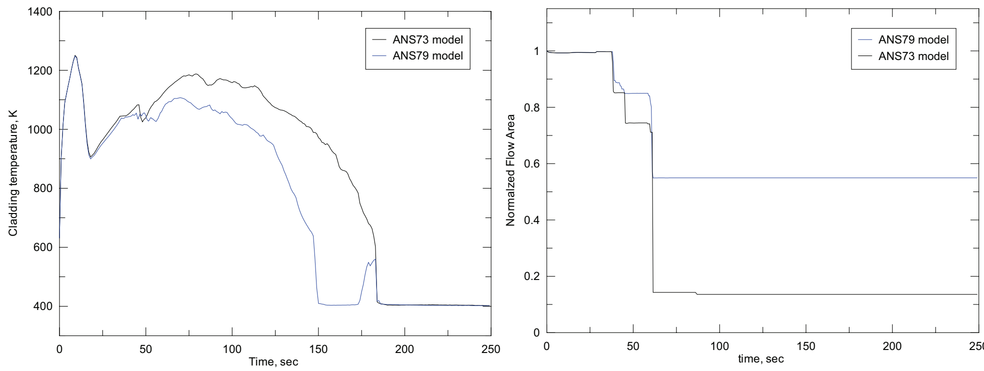
Figure 10 shows a comparison of cladding temperature for seven cases of flow blockage. The blockage level was adjusted by decreasing the multiplier to the number of fuel rods. As shown in the figure, the PCT during blowdown was the same for all the cases, while the PCT during reflood was tended to increase by the blockage level with the exception of the interval around 40% (see Fig. 10b). Moreover, a significant increase in PCT was found at around 80% blockage. The unexpected result in the reflood PCT at 43% blockage was considered due to the interaction of several code models and needs a further study. Based on the above findings, the effect of flow blockage due to swelling and rupture from the no blockage to 86% blockage can be 40 K in reflood PCT.

As discussed above, the use of conservative core decay heat model was considered as one of the reasons for such a high blockage. To confirm it, a calculation was conducted with



(a) Time trend (b) Reflood PCT

Figure 10: Cladding temperatures versus flow blockages.



(a) Cladding temperature (b) Hot channel blockage

Figure 11: Cladding temperatures and flow blockage for two decay heat models.

adopting ANS79 core decay heat model [4]. Figure 11 shows a comparison of cladding temperatures for two cases. As shown in the figure, a significantly low reflood PCT was predicted by the ANS79 model. The predicted flow blockage was shown in Fig. 11(b), which indicated only 45% blockage due to swelling and rupture. This means that the flow blockage due to the swelling and rupture is sensitive to the core decay power.

5 SUMMARY AND CONCLUSION

In the present study, a flow blockage modelling scheme was developed suitable for the calculation of large break LOCA using the system thermal-hydraulic code, MARS-KS. The modelling was setup such that the data of radial peaking factor and burnup of all the rods supplied by the core designer are divided into several sections. Moreover, the number of fuel rods corresponding to each section is determined with the combination of these two variables. The calculated cladding outer radii of the fuel rods in each section were used to calculate the level of flow blockage by introducing the servo valve model of the MARS-KS code. The present modelling scheme was applied to an analysis of a LBLOCA of an actual APR1400 NPP. The followings can be concluded:

1. Swelling and rupture were predicted for the fuel rods having a higher radial peaking factor and its magnitude was expanded by the level of fuel burnup. The level of the swelling and rupture was significant at the hot channel having a higher fluid temperature outside the fuel rods, i.e. higher hot channel peaking factor.
2. Flow blockage was predicted to be higher than 80% at the hot channel and the reflood PCT increased, in overall sense, by that level of blockage. The maximum impact on PCT increase was expected to be about 40 K for the range up to 86% blockage.

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